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NUCLEAR DESIGN DATA FOR EXPERIMENTS TO BE CONDUCTED IN HORIZONTAL THROUGH HOLE 1 OF THE PLUM BROOK REACTOR

By Robert G. Rohal, Patrick M. Finnegan, and E. J. Ciesla

Lewis Research Center
Cleveland, Ohio

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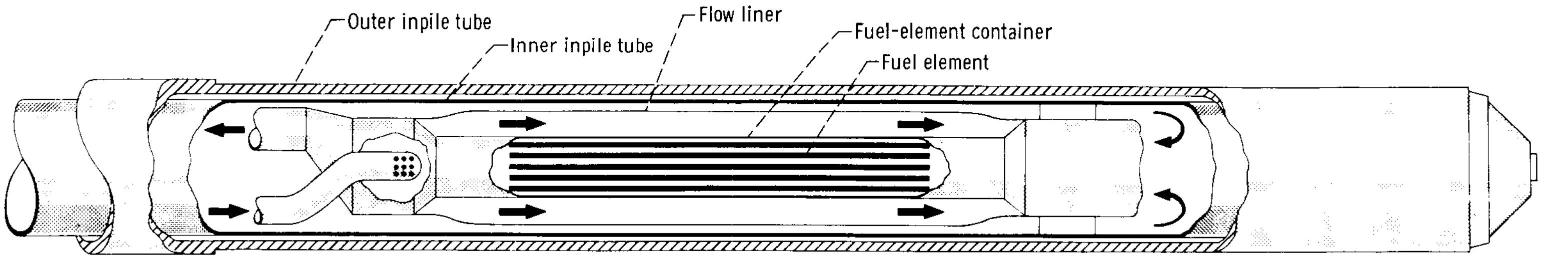
SUMMARY

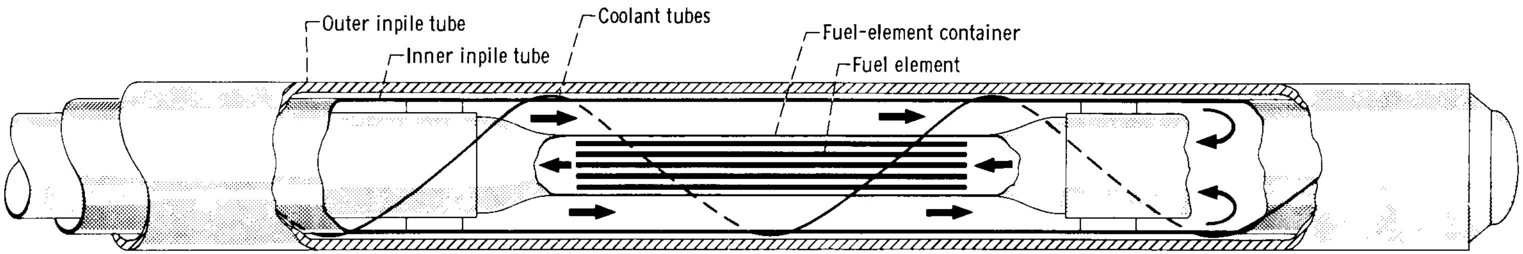
Three irradiation facilities have been designed for use in the Horizontal Through Hole 1 (HT-1) of the Plum Brook Reactor (PBR). These facilities are the Inpile Helium-Cooled Loop, the Single-Pass Argon-Cooled Facility, and the Inpile Capsule Facility. Experiments were conducted in HT-1 of the Plum Brook Mockup Reactor (MUR) to provide data for experiments employing these facilities. In these experiments, mockups of the three facilities were used to obtain the following information: (1) the thermal-neutron flux, which ranged from approximately 1×10^9 to 4×10^9 neutrons per square centimeter per second per kilowatt of reactor power along the axis of the inpile tubes; (2) the fission power in mockups of flat-plate fuel elements composed of tungsten and approximately 10 to 25 grams of fully enriched uranium, which ranged from approximately 0.04 to 0.10 watt per gram of enriched uranium per kilowatt of reactor power; (3) the gamma heating in the inpile tube structure, which ranged from approximately 0.5×10^{-4} to 1.0×10^{-4} watt per gram per kilowatt of reactor power; (4) the reactivity worth of the mock fuel elements and inpile tubes in both normal and accident conditions, which ranged from -4 to -40 cents, respectively.

INTRODUCTION

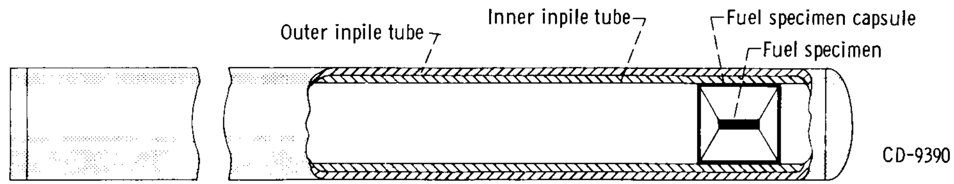
The irradiation facilities designed for use in Horizontal Through Hole 1 (HT-1) of the Plum Brook Reactor (PBR) provide a means of conducting various inpile irradiation experiments, such as fuel-element studies for advanced reactor concepts. Simplified sketches are given in figure 1 of these facilities (the Inpile Helium-Cooled Loop (ref. 1), the Single-Pass Argon-Cooled Facility, and the Inpile Capsule Facility). The nuclear



(a) Inpile helium-cooled loop facility.



(b) Single-pass argon-cooled loop facility.



(c) Inpile capsule facility.

Figure 1. - Irradiation facilities for horizontal through hole 1 of Plum Brook Reactor.

characteristics of the various inpile tubes and test specimens must be known in order to design irradiation experiments properly for the PBR, to tailor the test environment, and to ensure nonhazardous operating conditions. Low-power experiments were conducted by inserting mockups of the various inpile tube assemblies with and without mockups of various test specimens, into the HT-1 hole of the Mockup Reactor (MUR) at the Plum Brook Reactor facility. The mockup inpile tubes used are the helium inpile tube mockup, the argon inpile tube mockup, and the capsule inpile tube mockup. Two types of test specimen mockups, simulating two different fuel concentrations, were investigated.

The MUR experiments were conducted to provide four kinds of nuclear information:

(1) The neutron flux levels at various locations in the inpile tubes

(2) The fission power distributions produced in various fuel-element test specimens

(3) The reactivity worth of the inpile tubes with and without various test specimens

(4) The gamma heating distribution in the helium inpile tube mockup

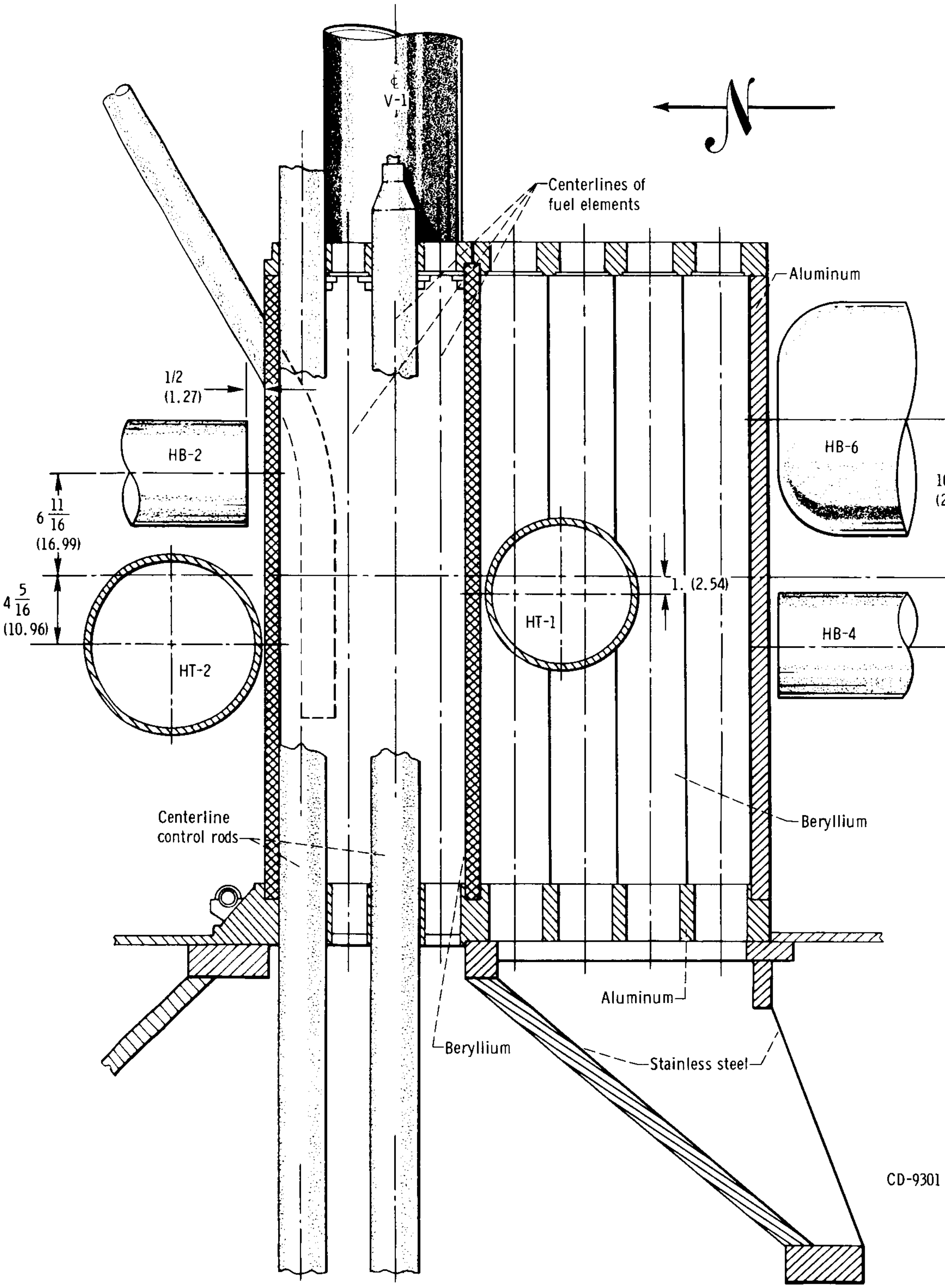
This information is intended to be used to design fuel-element experiments for the PBR irradiation facilities, to design improved inpile tube assemblies, to establish the effect of irradiation experiments on PBR reactivity and safety, and to aid in interpreting results of contemplated PBR tests.

DESCRIPTION OF APPARATUS AND TEST SPECIMENS

This section describes the MUR, the various inpile tube mockups, and the fuel-element test specimen mockups used in this study.

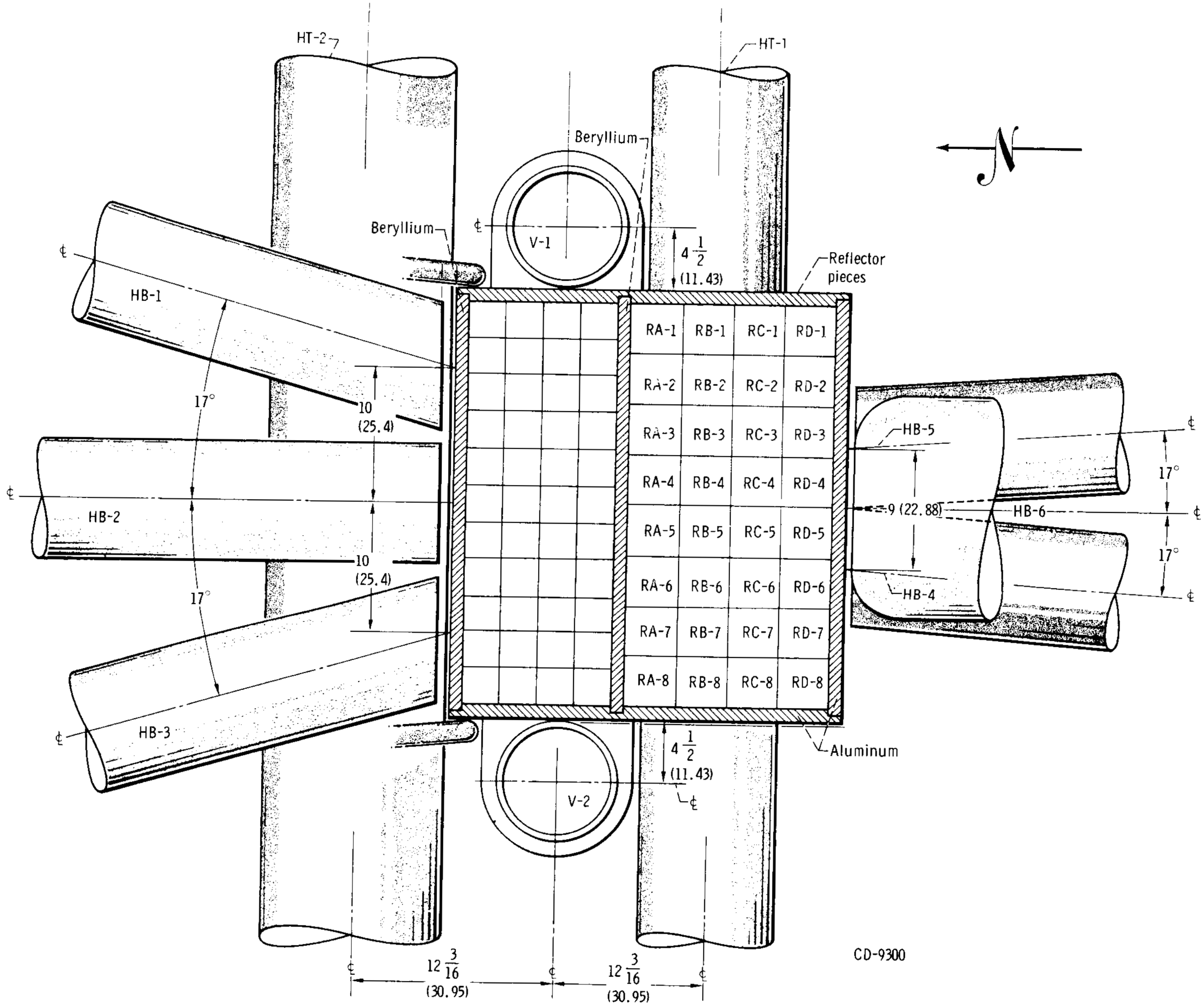
Mockup Reactor

The MUR at the PBR facility (fig. 2) is a low-power, swimming-pool-type reactor designed to simulate the PBR. The location of HT-1 (fig. 2(a)) is the same as that in the PBR. Provisions are made to load HT-1 by manipulating the inpile tube mockups from the surface of the pool. The fuel elements of the MUR are the same as the PBR fuel elements. When fully loaded, the MUR contains 27 fuel elements. The fuel elements are cooled by natural convection of light water which also serves as a moderator. The reactor core has a primary reflector of beryllium and a secondary reflector of light water. The core-loading patterns of the MUR used in these experiments are shown in figure 3. The loading pattern of figure 3(a) is termed "uniform" because all fuel elements except the fuel control rods have the same loading. The loading pattern of figure 3(b) is termed "mixed" because of the greater variety of uranium 235 (U^{235}) loading.

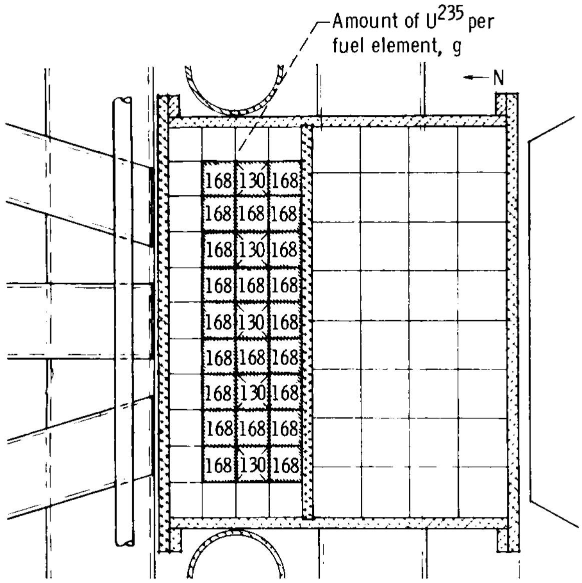


(a) Side view at core north-south vertical midplane.

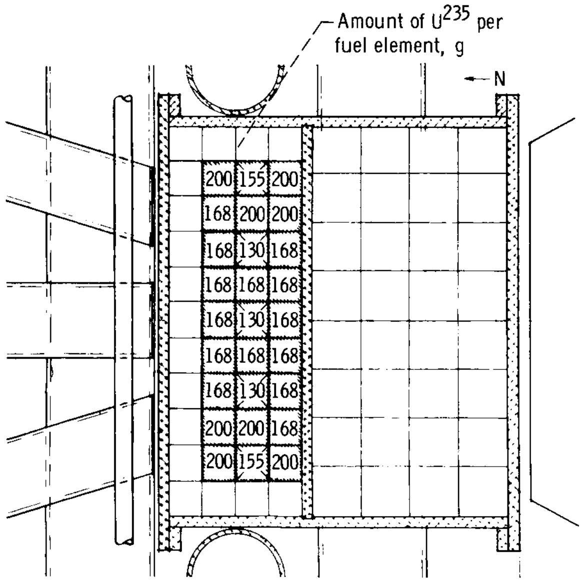
Figure 2. - Mockup reactor. (All dimensions are in inches (cm).)



(b) Top view.
Figure 2. - Concluded.



(a) With uniform core loading pattern.



(b) With mixed core loading pattern.

Figure 3. - Mockup reactor core. Crossed positions are control rods.

Inpile Tube Mockup Assemblies

Helium inpile tube mockup. - The mockup of the Helium-Cooled Loop inpile tube is shown in figure 4(a) and consists of an outer tube, an inner tube, and a test-element container. The outer tube is a sealed aluminum tube, $8\frac{1}{2}$ inches (21.6 cm) in diameter, about 4 feet (1.22 m) long, and with a wall thickness of 0.50 inch (1.27 cm). This mockup can simulate the outer pressure vessel of the Helium-Cooled Loop inpile tube (fig. 1). The end cap of the outer tube provides a handle for inserting the inpile tube mockup into the MUR. The inner tube is a 0.140-inch- (0.356-cm) thick stainless-steel container which simulates the inner pressure vessel of the Helium-Cooled Loop inpile tube. The end cap of the inner tube provides a shaft for axial positioning of the test-element container. The innermost container is a sealed aluminum box $24\frac{1}{2}$ inches or 62.2 centimeters long and with a 2- by 2- by $\frac{1}{8}$- inch or 5.08- by 0.317-centimeter wall, which holds the mockup fuel elements. The major difference between the mockup and the actual inpile tube is in the thickness of the inner and outer pressure vessels. In the actual inpile tube the outer pressure vessel is 0.474 inch (1.205 cm) thick and the inner pressure vessel is 0.112 inch (0.285 cm) thick. Also the actual helium inpile tube contains an 0.036-inch (0.0915-cm) stainless-steel flow liner which was not included in the helium inpile tube mockup.

Argon inpile tube mockup. - The Argon-Cooled Facility inpile tube mockup shown in figure 4(b) consists of an outer inpile tube can, an inner tube, and a test-element container. The outer tube is the same as the outer tube used for the helium inpile tube mockup. The inner tube is a 0.28-inch- (0.711-cm-) thick sealed stainless-steel container which simulates the inner pressure vessel of the Argon-Cooled Facility inpile tube. The test-element container is the same as the test-element container used for the helium inpile tube mockup. The major difference between the argon inpile tube mockup and the actual argon inpile tube is in the thickness of the inner and outer pressure vessels. In the actual inpile tube the outer pressure vessel is 0.360 inch (0.915 cm) thick and the inner pressure vessel has an effective thickness of 0.155 inch (0.394 cm). The actual inpile tube also contains a number of cooling tubes on the exterior of the inner tube which are not included in the inpile tube mockup.

Capsule inpile tube mockup. - The mockup of the Capsule Facility inpile tube is shown in figure 4(c). The mockup consists of an outer aluminum inpile tube and an inner stainless-steel capsule. The inpile tube is $3\frac{1}{2}$ inches (8.89 cm) in diameter and 30 inches (76.2 cm) long. The capsule (fuel-plate container) is $2\frac{1}{4}$ inches (5.72 cm) in diameter and 4 inches (10.16 cm) long. This mockup simulates the capsule inpile tube exactly.

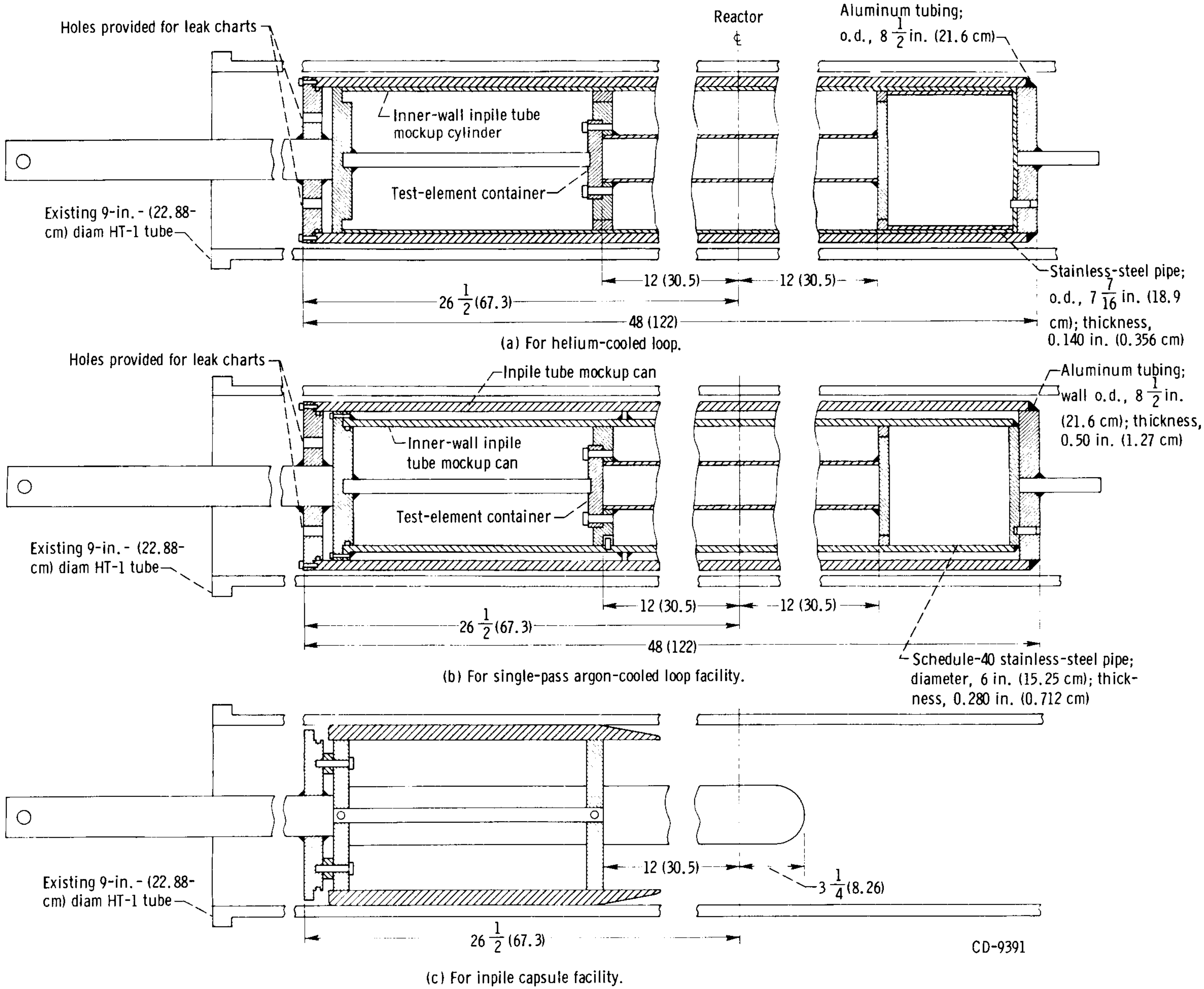
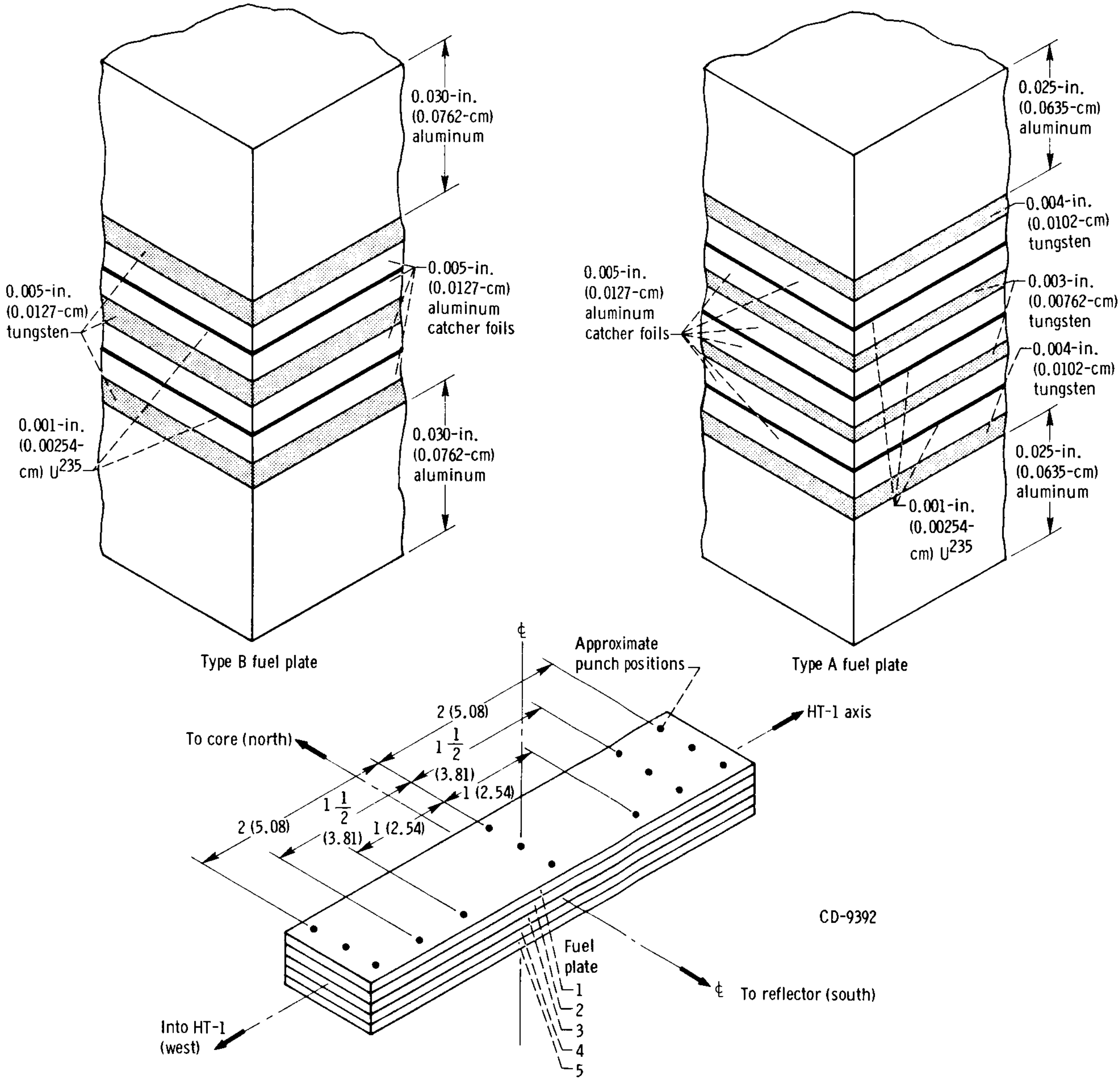


Figure 4. - Inpile tube mockups for three irradiation facilities. (Dimensions are in inches (cm).)



Five-plate fuel element showing approximate location of foils punched from aluminum catcher foils

Figure 5. - Typical fuel elements. (Dimensions in inches (cm).)

Fuel-Element Mockup Assemblies

Fuel elements were simulated by using aluminum, tungsten, and fully enriched uranium foils, as shown in figure 5. A typical fuel-element assembly, which consists of five fuel plates, is shown to indicate its orientation with respect to the MUR. The punch positions on the assembly show where disks were removed from the catcher foils. These disks were used to determine the fission power distribution throughout the mockup fuel elements. Two types of test specimen (A and B) were investigated, which represent tungsten - uranium dioxide ($W-UO_2$) fuel plates of different fuel concentrations. The enlargements in figure 5 show mockups of these test specimens. For each type of fuel plate, the outer layers of aluminum represent coolant flow channels between plates assembled into a fuel element.

Type A element. - The type A plate contains several layers of 0.001-inch- (0.00254-cm-) thick U^{235}, 0.001-inch- (0.00254-cm-) thick tungsten, and 0.005-inch- (0.0127-cm-) thick aluminum catcher foils, as well as 0.025-inch- (0.0635-cm-) thick aluminum. The total single-plate thickness is 0.097 inch (0.2465 cm), which yields a type A element (five-plate stack) thickness of about 1/2 inch (1.27 cm). The width and length are 1 inch (2.54 cm) and 4 inches (10.16 cm), respectively. The type A element contains approximately 19.5 grams of U^{235}.

Type B element. - The type B plate contains several layers of 0.001-inch- (0.0254-cm-) thick U^{235}, 0.005-inch- (0.0127-cm-) thick tungsten, and 0.005-inch- (0.0127-cm-) thick aluminum catcher foils, as well as 0.030-inch- (0.0762-cm-) thick aluminum. The total single-plate thickness is 0.097 inch (0.2465 cm), which yields a type B element (five-plate stack) thickness of about 1/2 inch (1.27 cm). The width and length are the same as for the type A element. The type B element contains approximately 12.5 grams of U^{235}.

EXPERIMENTAL METHODS

Thermal-Neutron Flux-Distribution Measurements

The thermal-neutron flux measurements in these experiments were made with 20-mil (0.508-mm) cadmium-covered and bare gold foils taped at specific positions in the inpile tube mockups. The gold foils were 5 mils (0.127 mm) thick and 1/4 inch (0.635 cm) in diameter. The irradiated bare and cadmium-covered gold foils were counted on a 512-channel pulse-height analyzer employing a sodium iodide crystal. The output of the analyzer in this case is the number of counts under the gold 198 (Au^{198}) photopeak at 0.411 MeV (7.88×10^{6} J). The analyzer count data, elapsed time from reactor scram to counting, foil data (foil material and foil mass), foil counting position relative to the detector, length of irradiation, and reactor power level are used as input for a computer program. The program calculates the absolute disintegration rate of each foil corrected to scram time and then converts this to a "flux per watt" number as an output. The flux per watt output values are then converted to thermal flux per watt values by correcting cadmium-covered data for thermal-neutron leakage through the covers, subtracting the corrected cadmium-covered data from the bare gold data, and applying a correction factor for flux depression by the detector.

Fission Power Distribution Measurements

The test-element power distribution was measured with aluminum catcher foils (ref. 2). The aluminum catcher foils (0.005 in. or 0.0127 cm thick) were placed adjacent to 0.001-inch- (0.00254-cm) thick U^{235} foils. The fission fragments of the surface atoms of U^{235} escape from the U^{235} foil and become lodged in the adjacent aluminum foil. Small disks (about 5/32 in. or 0.397 cm in diam) were then punched from the larger aluminum foil.

Relative power levels were obtained by counting the aluminum catcher disks in either the 512-channel pulse-height analyzer or the beta scintillation counters. The foils were counted relative to a standard foil to compensate for the decaying of the radioactive foils. This simplification was not available when counting in the analyzer. Therefore, a decay curve for the aluminum foils had to be constructed, and the foils were corrected for decay by using this curve. After correcting the counts for background (and decay time for the analyzer counts), the data gave relative levels of the fission heat generation throughout the plate.

The aluminum disks were calibrated in the following manner to establish the absolute power level. In a particular run, uranium disks directly adjacent to the aluminum disks were punched out. These uranium disks were analyzed, as described later in this section, to establish the average number of fissions per gram of fuel material. When aluminum foils directly adjacent to this uranium foil are counted, the count rate corresponds to a certain number of fissions per gram. Since the relative counts of the catcher foils are known, all that remains is to convert each relative count rate to an absolute heat generation. Once the reactor power, the time of irradiation, and the amount of energy given off with each fission are known, the number of fissions per gram of U^{235} can be expressed in terms of watts per gram of U^{235} per kilowatt of reactor power.

The number of fissions per gram in an irradiated U^{235} foil was determined by measuring the amount of the fission product lanthanum 140 (La^{140}) in the foil. The amount of La^{140} was found by two methods. In the first method the irradiated foil was allowed to decay for 5 days; the barium 140 (Ba^{140}) was then separated by radiochemical techniques. The La^{140} content was determined from the Ba^{140}. In the second method the La^{140} photopeak of the foil was counted after it had decayed for 5 days. Since the fission yield of La^{140} is known and the content of La^{140} can be found by either method, the number of fissions per gram can be determined. In this study both methods were used to determine the amount of La^{140}. The number of fissions per gram found by each method are in excellent agreement.

Gamma Heating Measurements

Gamma heat-generation rates were measured within the helium loop mockup. The measurements were made with chemical (ferrous sulfate) dosimeters which were unidirectionally alined at various positions throughout the mockup interior.

The ferrous sulfate dosimeter is the most precise of the liquid-chemical dosimeters. The precision of the dosimeters used for the measurements made within the helium loop inpile tube mockup is ±2 percent. Added to this uncertainty are other uncertainties such as predicting the correct reactor power, predicting a correct correspondence between the MUR results and the PBR results, and error in relating energy deposited in the ferrous sulfate to energy deposited in any other material (approx. 5 percent). These measurements and their corresponding map of gamma heat generation throughout the interior of the mockup are expressed in watts per gram per kilowatt of reactor power. These values are for energy absorption in water because the ferrous sulfate dosimeter has essentially the same mass energy absorption coefficient as water. However, the values of gamma heat generation at the various positions within the mockup may be used when materials other than water are present at the positions, if appropriate conversions of the data for other materials can be made. If other inpile tube configurations are used in conjunction with the data, the attenuation of the gamma flux by the configuration must be considered.

The MUR measurements may be used to estimate the gamma heating when the actual inpile tube is inserted in the PBR; however, the values should be corrected by a factor of 1 to 1.13 depending on the life of the PBR cycle. The factor increases with the life of the cycle, due to the delayed gamma field buildup with length of cycle (see ref. 3).

Reactor Power Measurements

Since the results of the experiment are directly influenced by the reactor power, it is imperative that as accurate a power measurement as possible be made. A power calibration must be made for each core configuration in which experiments are to be run.

This power calibration is accomplished by the irradiation of bare gold foils in the core. Vertical traverses are made in at least four fixed fuel elements, and a minimum of two foils each are placed in the remainder of the fixed elements. From the irradiated foil data and a knowledge of the gold-cadmium ratio in the core, an average power is determined for each element. The power generated in each element is then summed to give the average reactor power, exclusive of the power generated in the fueled sections of control rods. This power is then increased by a factor to account for power generation in control-rod fuel sections.

At the time a power calibration was made, bare gold wires and foils were also irradiated elsewhere in the reactor for future reference as to the power level. The bare gold wires were placed in the corner holes of reflector pieces RA-4, RB-4, RC-4, and RD-4, and the bare gold foils were placed on the perimeter of a rotating wheel in vertical test hole 1 (V-1) (fig. 2). From the irradiated wire and foil data and the measured power level, an activity per irradiation time per watt value is generated for each wire and foil location. The wire and foil activity in future runs is then used to determine the power level of those runs. In addition to monitor detectors placed in the reactor, the log N channel of the reactor instrumentation provides information as to the reactor power level. The channel consists of a gamma-compensated ion chamber located on the north side of the reactor below the level of the active core. Readout of the channel for a particular run, when compared with its readout for the power calibration run, yields the reactor power.

Reactivity Effects Measurements

The effect of the test element on core reactivity was measured in the following manner. With HT-1 flooded, the reactor was made critical, and level control-rod bank height was noted. With an inpile tube mockup in HT-1 the reactor was made critical, and level control-rod height was noted. The difference in control-rod height times an average control-rod reactivity worth per inch of travel between the two heights gives the inpile tube mockup worth in HT-1. The reactivity worth of the test element is the difference between the worth of the inpile tube mockup without an element in HT-1 and the worth of the inpile tube mockup with an element in HT-1.

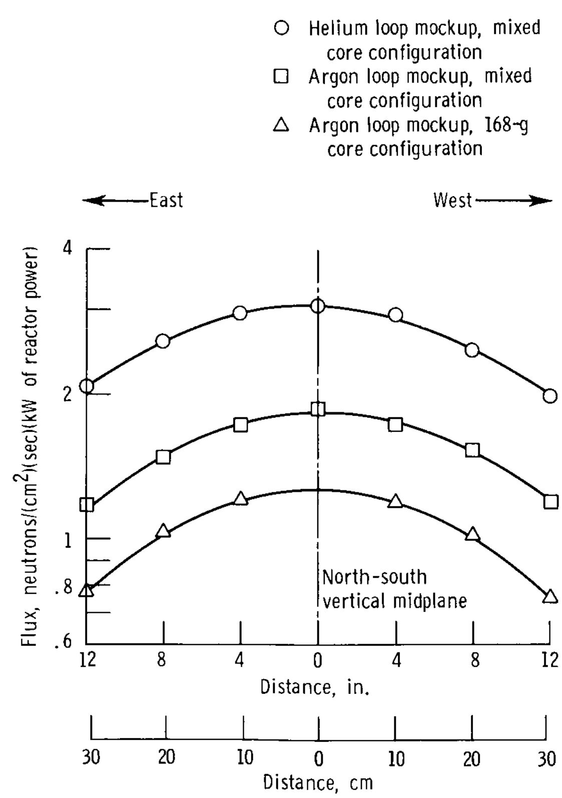
RESULTS AND DISCUSSION

The results of the MUR experiments are grouped primarily by the kinds of data (i.e., thermal flux distributions, fission power distributions, gamma heating distributions, and reactivity effects) and are presented in that order. Within each kind of data, the results are separated according to inpile tube configuration, that is, helium inpile tube mockup, argon inpile tube mockup, and capsule inpile tube mockup, in that order.

Thermal-Neutron Flux Distributions

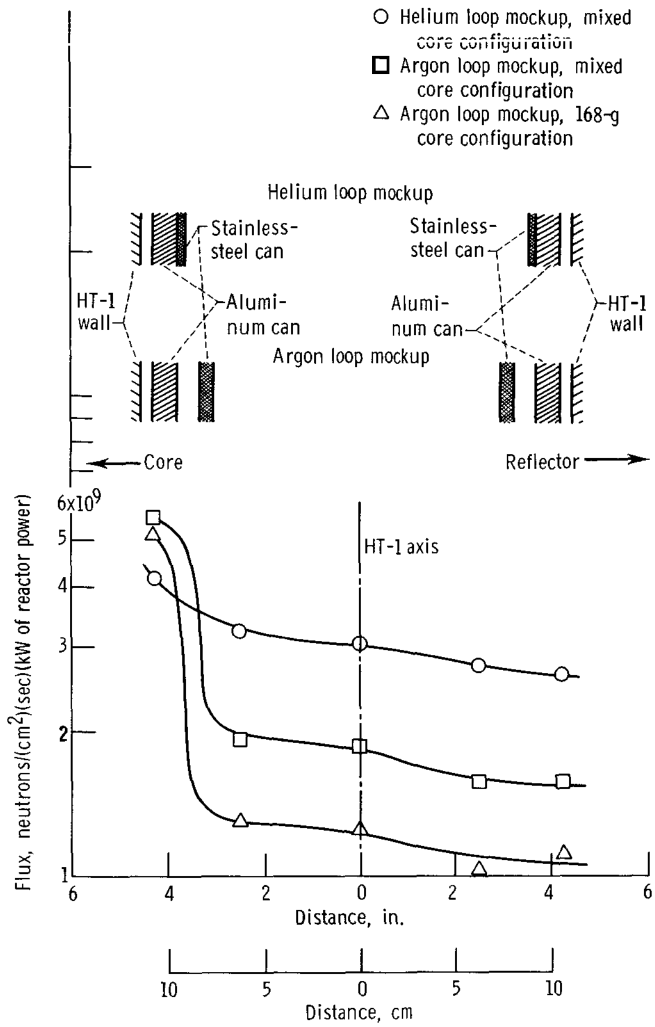
Helium inpile tube mockup. - The thermal-neutron flux distribution data for the helium inpile tube mockup are shown in figure 6. The axial flux distribution (fig. 6(a)) was taken along the HT-1 axis of MUR, while the transverse flux distribution (fig. 6(b)) was taken across the horizontal diameter of HT-1 at the north-south vertical midplane (fig. 2(b)). Both traverses were taken when the MUR was loaded with a mixed core configuration (fig. 3(b)).

The axial flux has a cosine distribution with a maximum value of 3.02×10^9 neutrons per square centimeter per second per kilowatt of reactor power. The axial flux is symmetric about the north-south vertical midplane. The transverse flux falls off gradually as the distance from the core increases.



(a) Along HT-1 axis with various loop mockups inserted.

Figure 6. - Thermal-neutron flux.



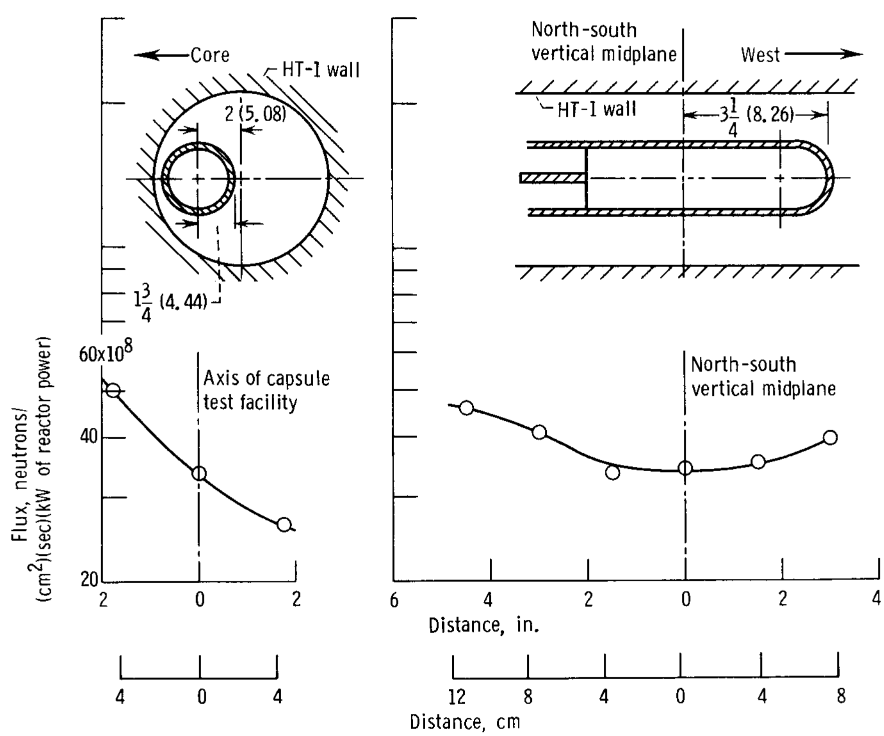
(b) Along horizontal diameter of HT-1 at the reactor north-south vertical midplane, with various loop mockups inserted.

Figure 6. - Continued.

Argon inpile tube mockup. - The thermal-neutron flux distribution data for the argon inpile tube mockup are also shown in figure 6. As before, the axial flux distribution (fig. 6(a)) was taken along the HT-1 axis, while the transverse flux distribution (fig. 6(b)) was taken across the horizontal diameter of HT-1 at the north-south vertical midplane (see fig. 2(b)). Two transverse and two axial distributions appear in figures 6(a) and (b) for the argon inpile tube mockup. One set of curves represents data for the MUR core loaded in a 168-gram configuration (see fig. 3(a)), and the other set of curves represents data for the MUR core loaded in a mixed core configuration (see fig. 3(b)). A comparison of these curves shows that the thermal-neutron flux level for a mixed core is approximately 35 percent higher than the flux in the uniform-loaded core configuration.

As before, the axial flux has a typical cosine distribution with a maximum value of 1.83×10^9 neutrons per square centimeter per second per kilowatt of reactor power for the mixed core configuration and 1.26×10^9 neutrons per square centimeter per second per kilowatt of reactor power for the 168-gram-core configuration. Both of these traverses are considerably lower than the traverse for the flux in the helium inpile tube mockup. The transverse traverses, as in the helium inpile tube mockup case, fall off as the distance from the core increases. The reduction in flux is greater for the argon inpile tube mockup because of the greater amount of absorber (stainless steel) in this inpile tube mockup. The two transverse traverses show similar separation which appeared on the axial plot because of the core loading.

Capsule inpile tube mockup. - The thermal-neutron flux distribution data for the capsule inpile tube mockup are shown in figures 6(c) and (d). The axial flux distribution (fig. 6(d)) was taken along the capsule inpile tube mockup axis, which is displaced horizontally 2 inches (5.08 cm) toward the core from the HT-1 axis. The transverse flux



(c) Along horizontal diameter of HT-1 at reactor north-south vertical midplane with capsule facility inserted. Mixed core configuration.

(d) Along axis of capsule facility inserted in HT-1. Mixed core configuration.

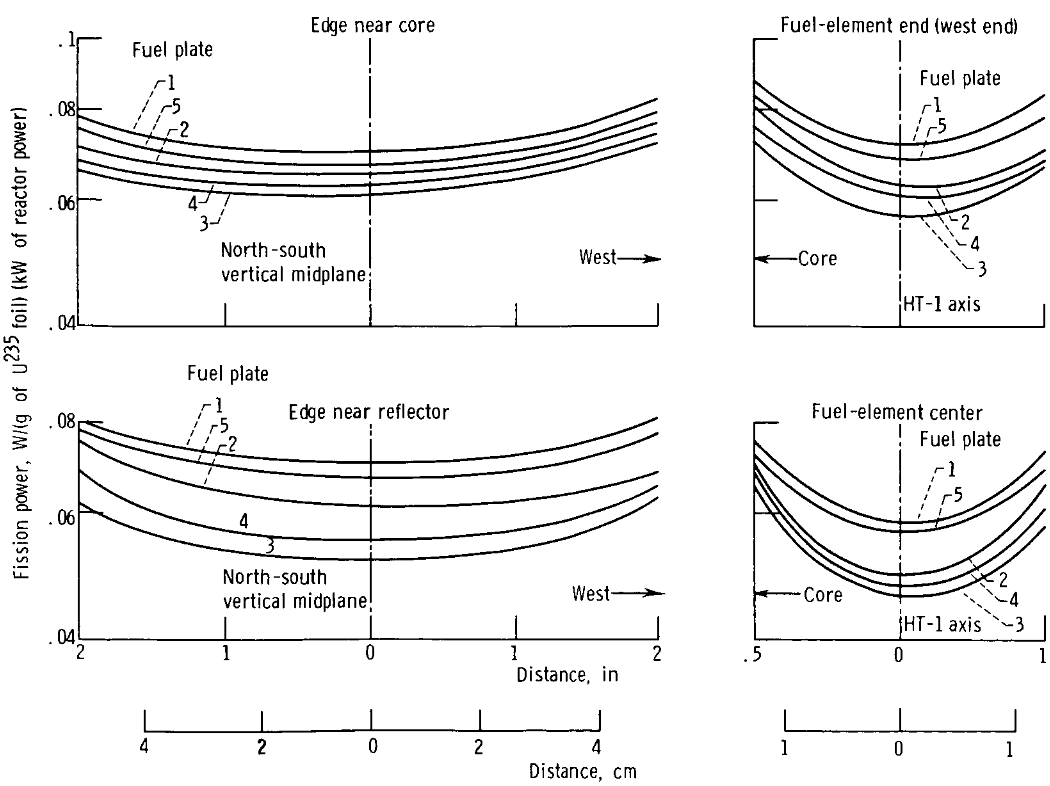
Figure 6. - Concluded. (Dimensions are in inches (cm).)

distribution (fig. 6(c)) was taken across the horizontal diameter of HT-1 at the north-south vertical midplane (see fig. 2(b)). Both traverses were taken when the MUR was loaded with a mixed core configuration (see fig. 3(b)).

The axial flux is symmetrical about the north-south vertical midplane. The transverse flux, as in the helium and argon inpile tube mockups, falls off as the distance from the core increases along the horizontal diameter. The flux at the north-south vertical midplane at the capsule axis is 3.37×10^{9} neutrons per square centimeter per kilowatt of reactor power. This flux is about 10 percent higher than the flux at the helium inpile tube axis at the core north-south midplane.

Fission Power Distributions

Helium inpile tube mockup. - Fission power distribution data for various fuel-element configurations placed in the test-element container (fig. 4(a)) are shown in figures 7 to 13. The element is so placed in the inpile tube mockup that its axis coincides



(a) Along lines parallel to HT-1 axis.

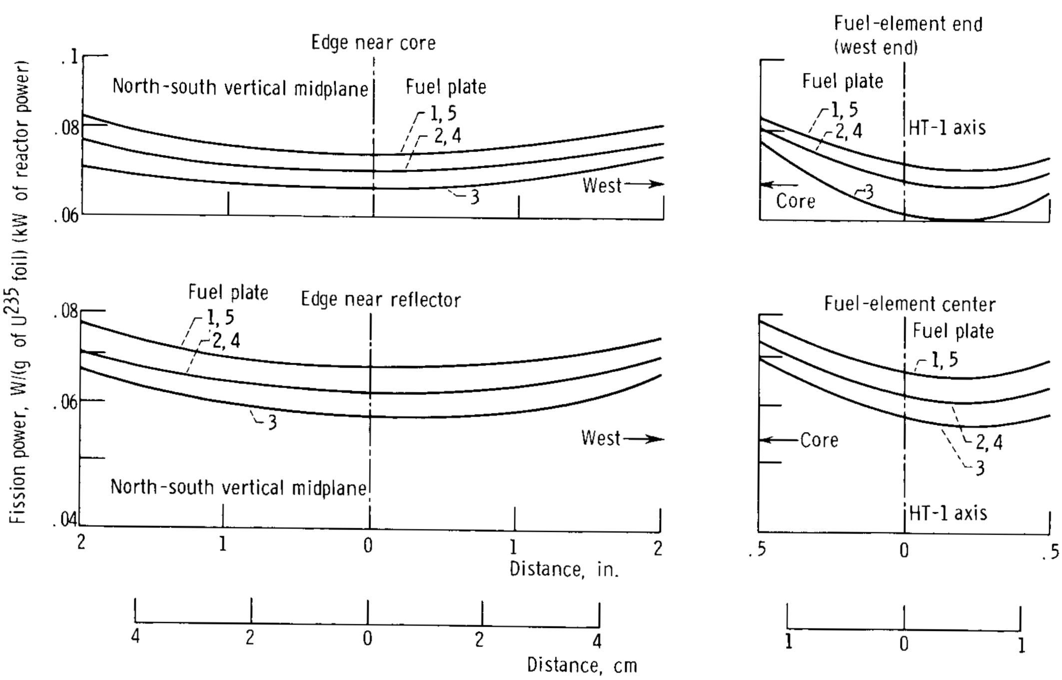
(b) Along lines parallel to horizontal diameter of HT-1.

Figure 7. - Fission power distribution in type A fuel element inserted in helium loop mockup.

with the inpile tube axis and its vertical midplane (perpendicular to the element axis) coincides with the core north-south vertical midplane (fig. 5). Figure 5 also shows the relative location of the catcher foils. Each plate of the element is referred to by a number with 1 denoting the top plate, 2 denoting the plate directly below the top plate, etc..

Each figure consists of four traverses through the element for each of the five fuel plates (for alternate plates in fig. 11). The two traverses titled "edge near core" and "edge near reflector" are parallel to the axis of HT-1 and represent data obtained along the reflector and core edges of the element. The two traverses titled "fuel-element end" and "fuel-element center" are parallel to the HT-1 horizontal diameter and represent data obtained along the west edge and center of the element. All the experiments described in this section were performed with the MUR core in a mixed configuration (see fig. 3(b)).

The basic element used in these studies was composed of five fueled plates with each plate 1 inch (2.54 cm) wide by 4 inches (10.16 cm) long. Two fuel-element loadings were examined, type A (heavier loaded) and type B. Figure 7 shows the fission power resulting when the type A element is placed in the helium inpile tube mockup. The fission power decreases nearer the center of the element. This "power depression" is a common occurrence in fuel elements and is a result of the self-shielding effects of the fuel. In figure 7(b), where the traverses are parallel to HT-1 horizontal diameter, it is apparent



(a) Along lines parallel to HT-1 axis.

(b) Along lines parallel to horizontal diameter of HT-1.

Figure 8. - Fission power distribution in type B fuel element inserted in helium loop mockup.

that the core side of the element has a higher fission power production than the reflector side. The average fission power in this element was 0.063 watt per gram of uranium per kilowatt of reactor power. The average fission power in all the mockup fuel elements was obtained by first setting up a three-dimensional grid for the element. In all cases the grid contained 16 divisions along the length of the element, 8 divisions along the width, and 4 divisions along the depth. Then the power at each grid point was obtained from the fission power curves (e.g., fig. 7), and the average fission power was the average of the fission power of all grid points. The peak-to-average power for this element is 1.38.

In figure 8 the fission power distribution resulting when a type B element is placed in the helium inpile tube mockup is shown. This distribution is similar to the distribution of the type A element except the average power is slightly higher at 0.067 watt per gram of uranium per kilowatt of reactor power; and the peak-to-average power is slightly lower at 1.23.

When experiments in the helium and argon inpile tubes are carried out at high tem-

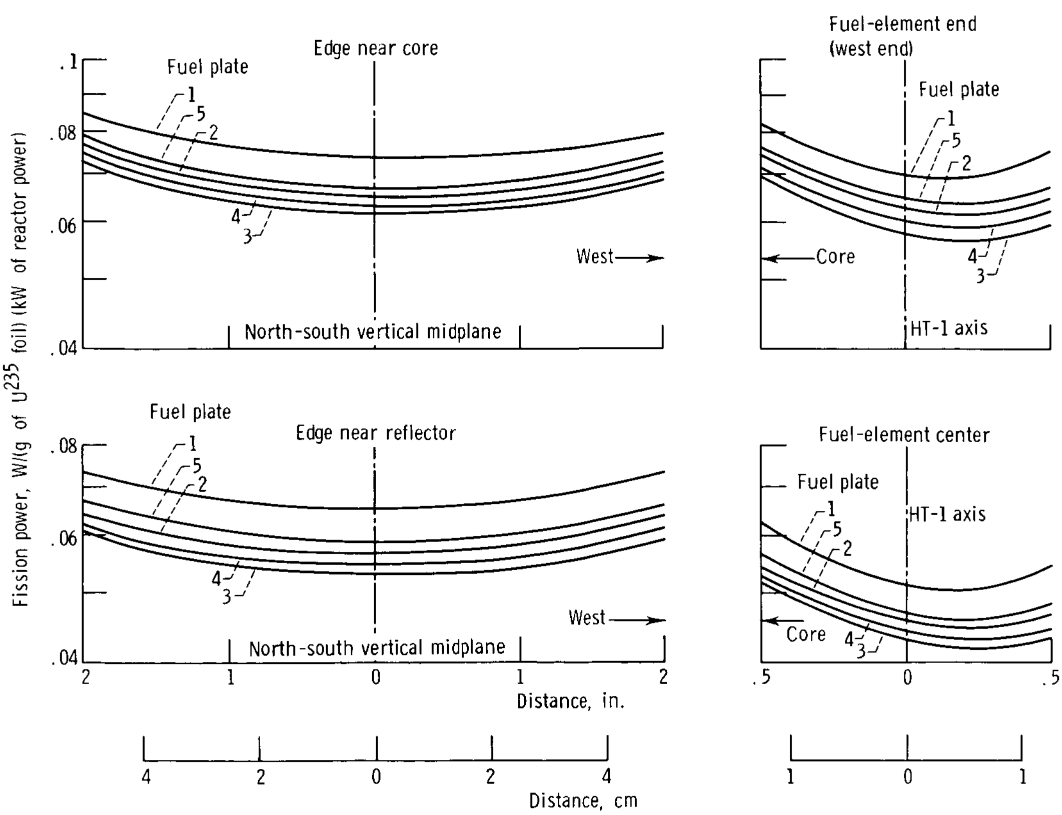
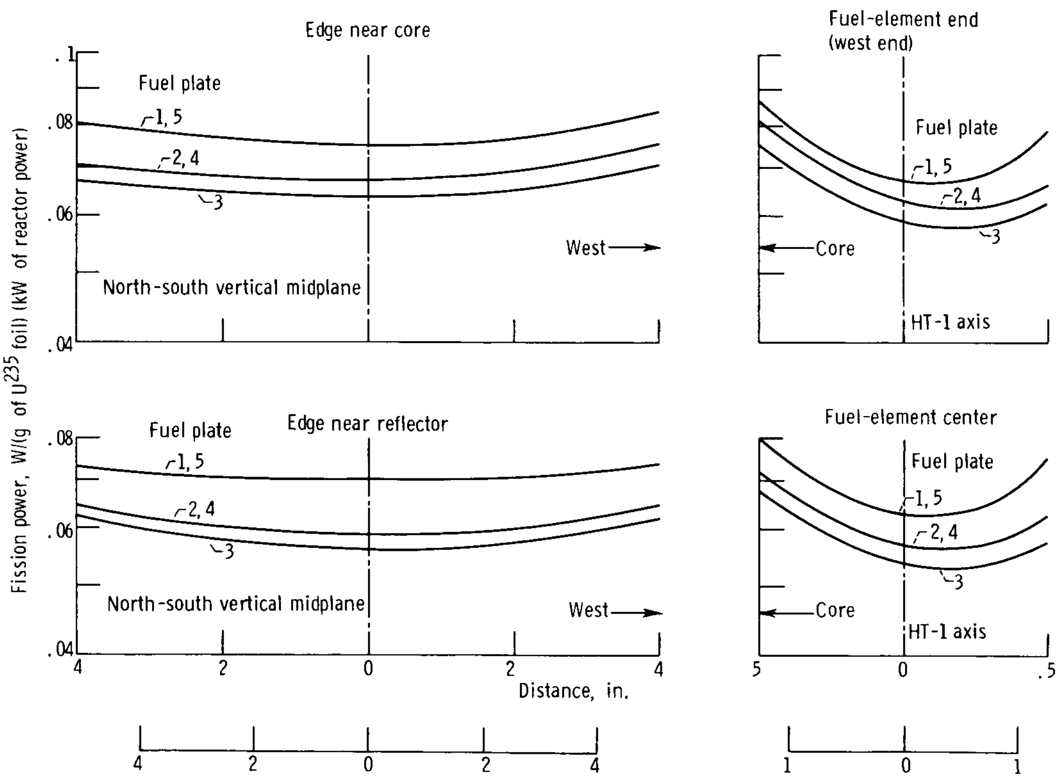


Figure 9. - Fission power distribution in type B fuel element enclosed in tungsten-molybdenum box and inserted in helium loop mockup.

peratures, the structure of the inpile tube must be protected from the thermal radiation of the irradiated fuel element. A box, which contains the fuel element under such conditions, has been designed to provide adequate protection for the inpile tube. The box is made of 0.020-inch- (0.051-cm-) thick tungsten inside an 0.060-inch- (0.153-cm-) thick molybdenum box. The fission power distribution in the fuel element placed in such a box is shown in figure 9. A comparison of figures 8 and 9 shows the effect of placing the tungsten-molybdenum box around the type B element. The distribution without the box is slightly higher, and the power depression without the box is slightly less. The average power is 0.062 watt per gram of uranium per kilowatt of reactor power and the peak-to-average power is 1.26.

The effect on the fission power distribution of changing the fuel-element length is shown in figure 10. The length of the type B element was doubled, from 4 to 8 inches (10.16 to 20.32 cm). The general appearance of the distributions is the same for the distributions of figure 8. However, the average power decreased from 0.067 watt per gram of uranium per kilowatt of reactor power to 0.066 watt per gram of uranium per kilowatt of reactor power when the length of the type B element was doubled. The peak-



(a) Along lines parallel to HT-1 axis.

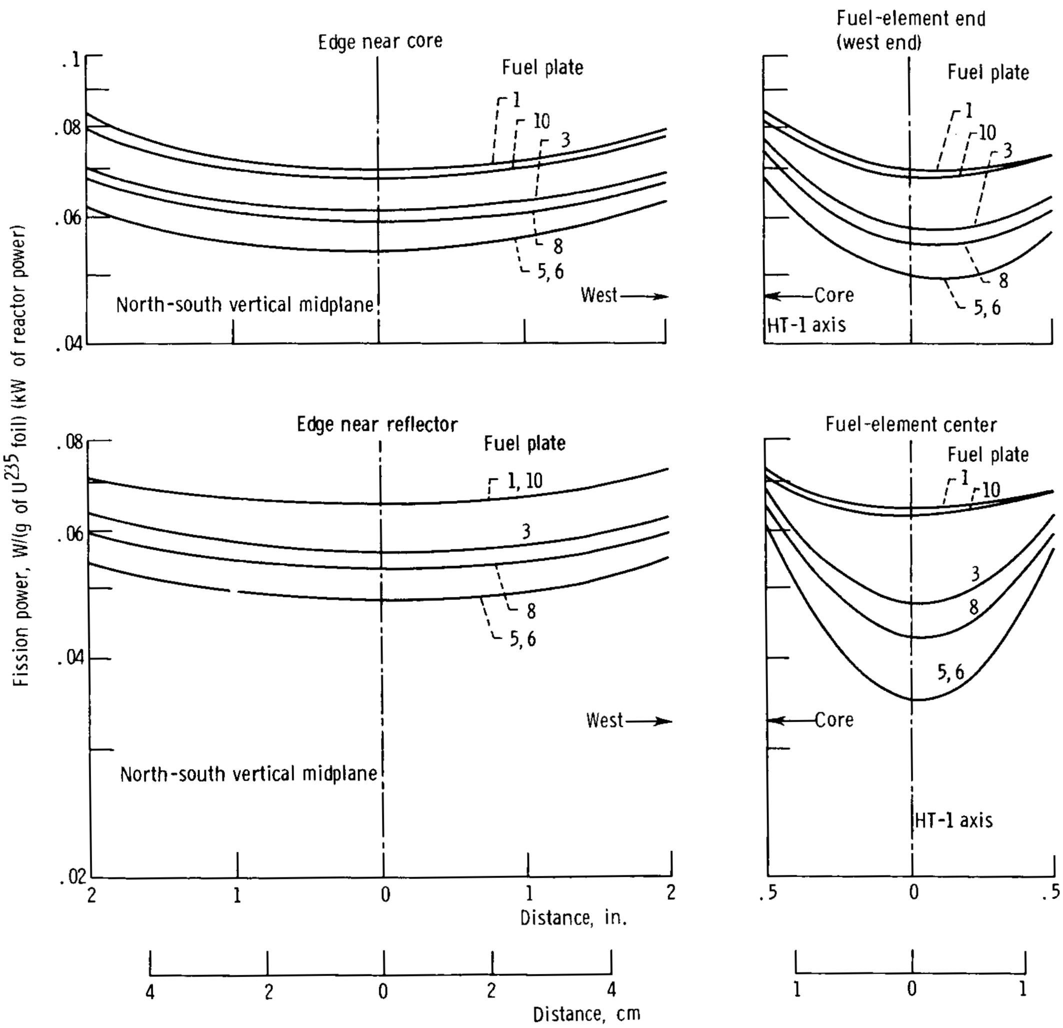
(b) Along lines parallel to horizontal diameter of HT-1.

Figure 10. - Fission power distribution in two type B fuel elements (in series) inserted in helium loop mockup.

to-average power increased from 1.23 to 1.33 when the length of the type B element was doubled.

The effect on the fission power distribution of increasing the number of plates in the type B element from 5 to 10 is shown in figure 11. No catcher foil measurements were taken on plates 2, 4, 7, and 9. Again the general appearance of the distributions is the same as for the distributions of figure 8. The average power decreased from 0.067 to 0.055 watt per gram of uranium per kilowatt of reactor power, and the peak-to-average power increased from 1.23 to 1.55.

During a reactor cycle, the control-rod position can vary over a wide range. The level control-rod bank height was changed from 21.2 to 23.6 inches (53.8 to 60.0 cm) to



(a) Along lines parallel to HT-1 axis.

(b) Along lines parallel to horizontal diameter of HT-1.

Figure 11. - Fission power distribution in two type B fuel elements (in parallel) with elements in helium loop mockup.

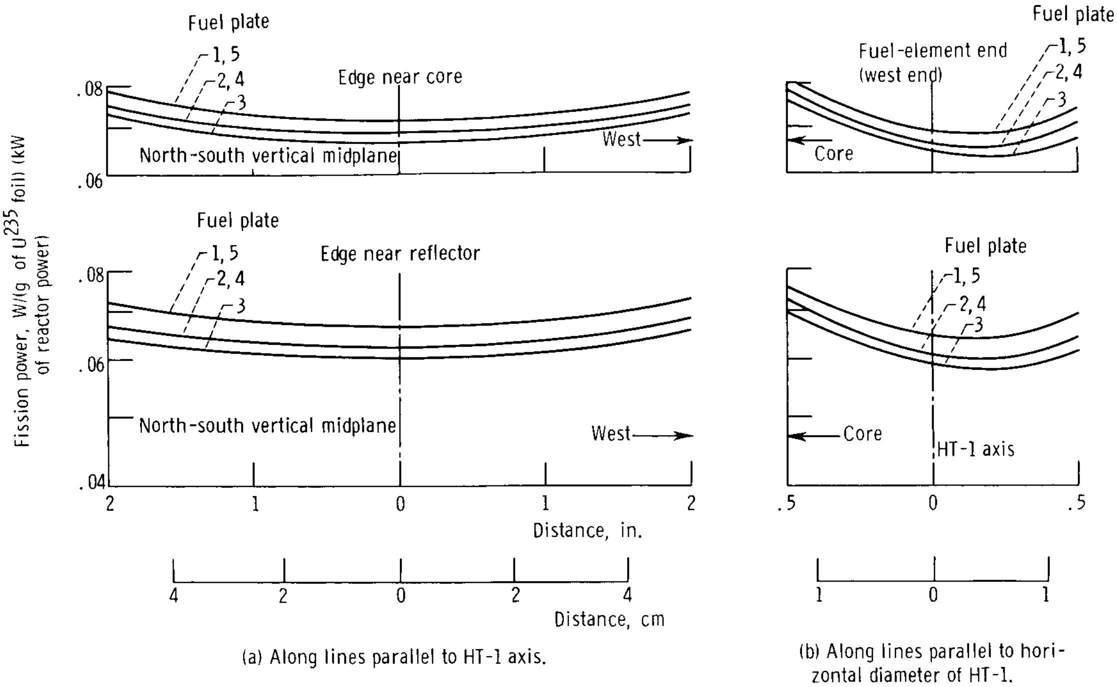
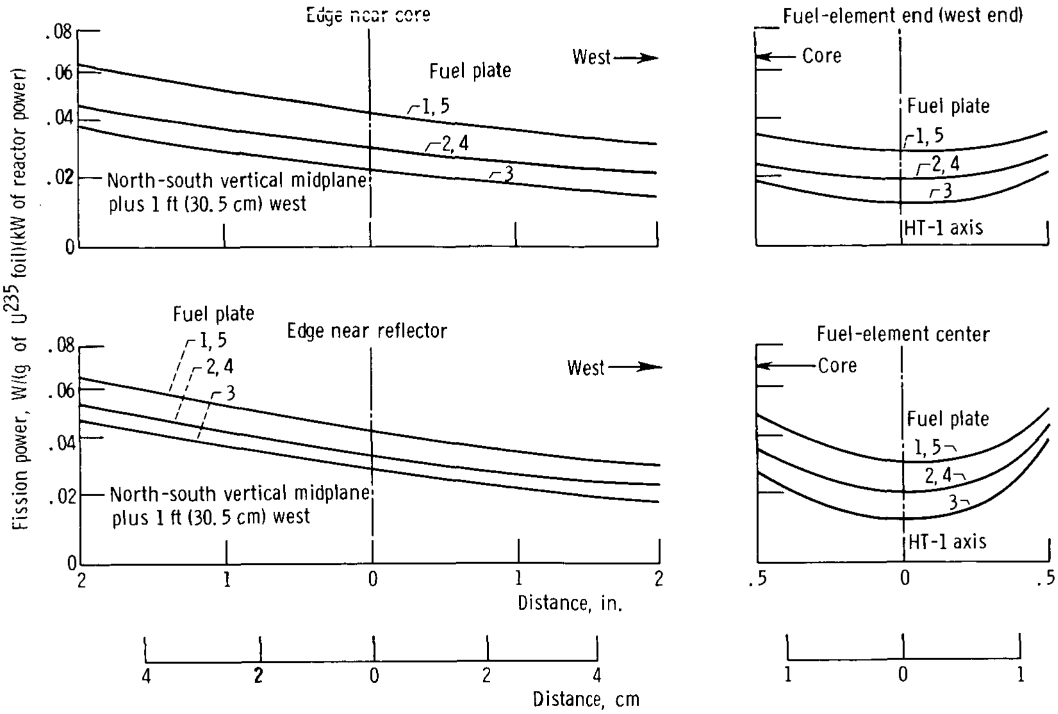


Figure 12. - Fission power distribution in one type B fuel element with element in helium loop mockup and core poisoned down.

investigate the effect of such control-rod variation on the fission power distribution in a type B element. The results of this investigation are shown in figure 12. The general appearance of the distributions is the same as the distributions of figure 8. The average power decreased from 0.067 to 0.064 watt per gram of uranium per kilowatt of reactor power, and the peak-to-average power increased from 1.23 to 1.26.

Fission power distribution data for a hypothetical accident condition are given in figure 13. The accident is that of the element holder failing and the element being pushed to one end of the helium mockup inpile tube by the coolant flow. A type B element was so placed in the helium inpile tube that the element's vertical midplane (the plane perpendicular to the element axis) was 12 inches (30.5 cm) to the west of the core north-south vertical midplane. The general appearance of the transverse distributions is the same as the distributions of figure 8. The axial traverses appear tilted with the lower power at the west end of the element. The average power for the displaced element was 0.054 watt per gram of uranium per kilowatt of reactor power, while the average power for the normally positioned element was 0.067 watt per gram of uranium per kilowatt of reactor power. The peak-to-average power for the displaced element was 1.33 as compared with 1.23 for the normal positioned element.

<u>Argon inpile tube mockup.</u> - An experiment was conducted in the argon inpile tube mockup to determine how the fission power distribution varies when an element taken



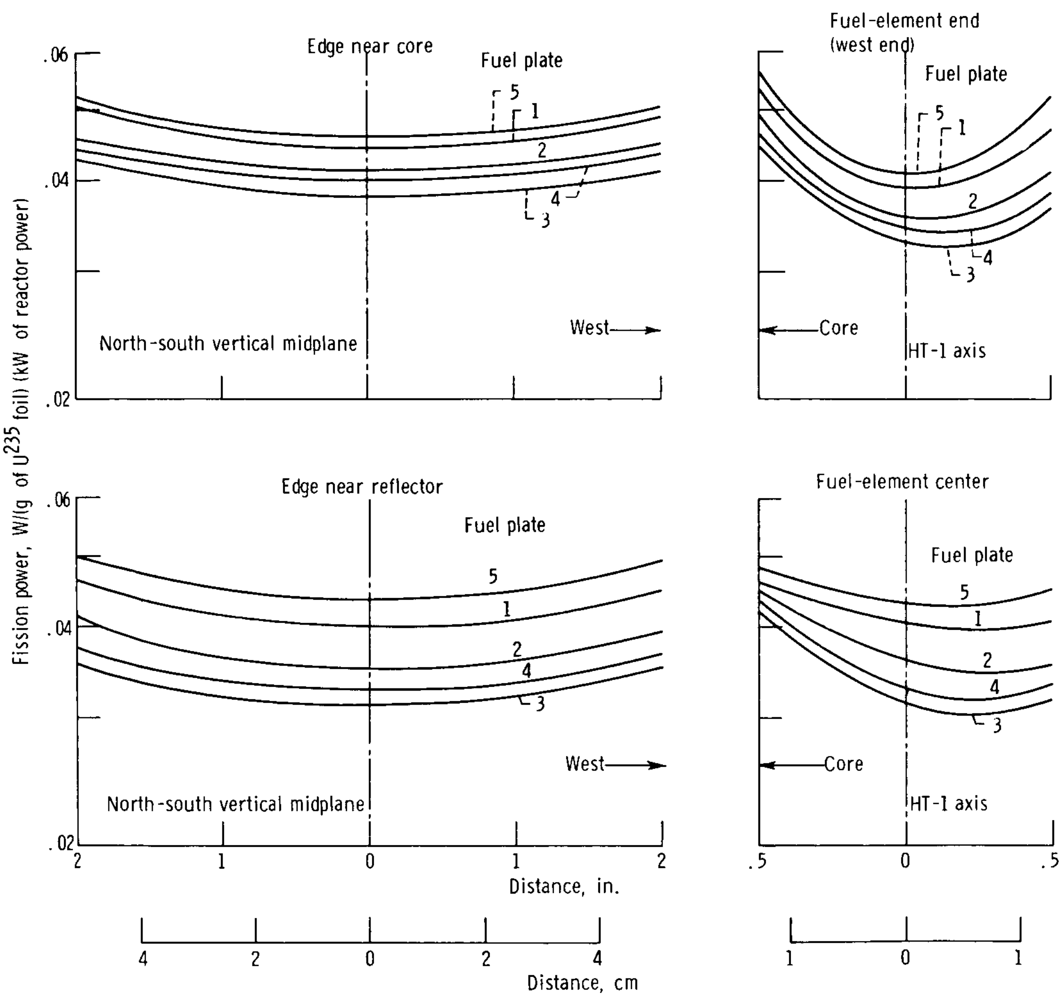
(a) Along lines parallel to HT-1 axes.

(b) Along lines parallel to horizontal diameter of HT-1.

Figure 13. - Fission power distribution in one type B fuel element with element in helium loop mockup and displaced 12 inches (30.5 cm).

from the helium inpile tube mockup is placed in the argon inpile tube mockup. The general appearance of the distributions shown in figure 14 appears to be the same as that of the distributions shown in figure 7. However, the average power for the type A element in the argon inpile tube mockup was 0.041 watt per gram of uranium per kilowatt of reactor power as compared with 0.063 watt per gram of uranium per kilowatt of reactor power for the same element in the helium inpile tube mockup. The peak-to-average power in the argon inpile tube experiment was 1.41 as compared with 1.38 in the helium inpile tube mockup.

Capsule inpile tube mockup. - The capsule test facility was designed to test small portions of fuel plates. The results of placing a 1-inch-long- by 7/8-inch-wide (2.54- by 2.22-cm) type B plate in the capsule inpile tube mockup are shown in figure 15. The distribution along lines parallel to the HT-1 axis is shown in figure 15(a) and the distribution along lines parallel to the horizontal diameter of the capsule inpile tube mockup in figure 15(b). The average fission power was 0.115 watt per gram of uranium per kilowatt of reactor power, and the peak-to-average power was 1.13.



(a) Along lines parallel to HT-1 axis.

(b) Along lines parallel to horizontal diameter of HT-1.

Figure 14. - Fission power distribution in one type A fuel element with element in argon loop mockup.

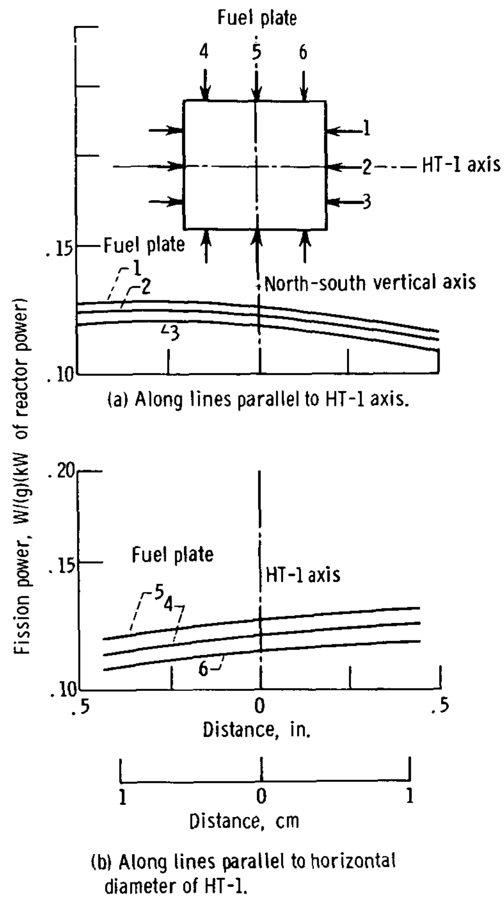
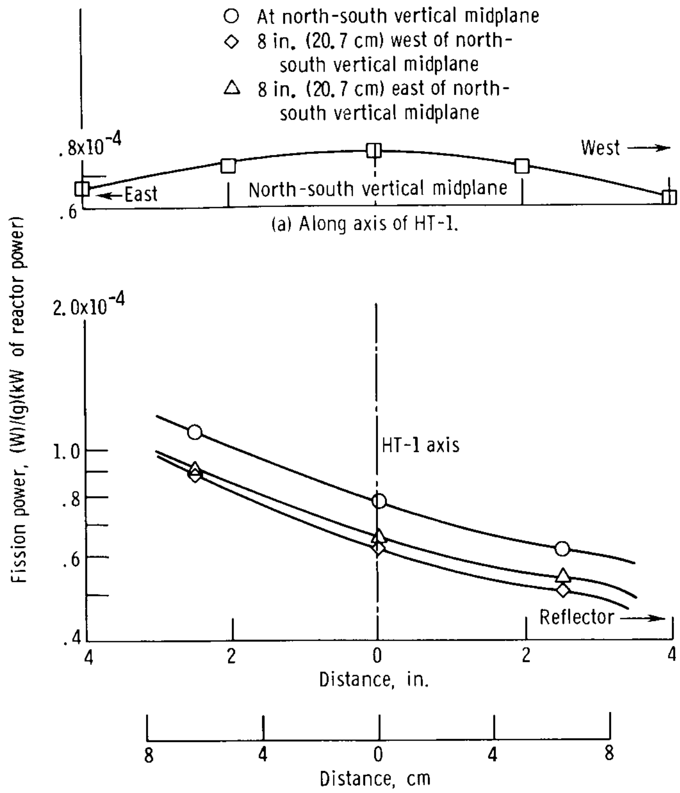


Figure 15. - Fission power distribution in type B fuel plate with plate in capsule facility mockup.

Gamma Heating Distributions

Gamma heating data were obtained for the helium inpile tube mockup only. The results are shown in figure 16. The axial distribution (fig. 16(a)) was taken along the HT-1 axis, and the transverse distributions (fig. 16(b)) were taken along the horizontal diameter of HT-1 in three vertical planes, the reactor north-south vertical midplane and 8 inches (20.32 cm) on either side of the midplane.

The maximum gamma heating occurring along the axis of HT-1 with the helium inpile tube mockup inserted is 0.78×10^{-4} watt per gram of uranium per kilowatt of reactor power. The axial gamma heating curves follow the shape of the axial thermal flux. The gamma heat generation ranged from approximately 0.5×10^{-4} to 1.0×10^{-4} watt per gram per kilowatt of reactor power.



(b) Across horizontal diameter of HT-1 in helium loop mockup.

Figure 16. - Ferrous sulfate gamma heating data.

Reactivity Effects

Reactivity worth data were obtained first for the various inpile tube mockup assemblies without fuel elements in the test-element container. Secondly, reactivity measurements were obtained for the various inpile tube mockup assemblies with fuel elements in the test-element container. These reactivity results are presented in table I.

The more pertinent results are the following. The reactivities of the helium, argon, and capsule inpile tube mockups are -40.2, -20.2, and -4.4 cents, respectively. As U^{235} is added to the inpile tube mockups, the reactivity approaches zero. If the accident of fuel-element displacement were to occur, the reactivity would change from -38.8 to -40.2 cents; a change of -1.4 cents. If the accident of inpile tube rupture (flooding)

TABLE I. - HORIZONTAL THROUGH HOLE 1 SURVEY EXPERIMENTAL RESULTS FOR MIXED CORE

Inpile tube mockup	Element type	Run condition	U^{235} weight, g	Rod height				Reactivity, cents	Average power, W/(g of U)(kW)	Peak-to-average power
				Zero run		Actual run				
				in.	cm	in.	cm			
Helium	--	No element	-----	------	------	------	------	-40.2	-----	----
Argon	--	No element	-----	------	------	------	------	-20.2	-----	----
Capsule	--	No element	-----	------	------	------	------	-4.4	-----	----
Helium	A	Flooded	18.65	21.148	53.716	21.289	54.074	-18.2	-----	----
Helium	A	Normal	19.23	21.145	53.708	21.444	54.468	-38.7	0.063	1.38
Helium	B	Normal	12.58	21.162	53.751	21.461	54.511	-38.8	.067	1.23
Helium	B	Tungsten and molybdenum	13.06	21.168	53.767	21.469	54.531	-38.9	.062	1.26
Argon	A	Normal	19.94	21.226	53.914	21.367	54.272	-18.1	.041	1.41
Helium	2B	Normal elements parallel	25.74	21.183	53.805	21.503	54.618	-37.6	.055	1.53
Helium	2B	Normal elements in series	25.33	21.214	53.884	21.492	54.590	-36.1	.066	1.33
Helium	B	Rod height changed	12.20	23.565	59.855	23.943	60.815	-34.0	.064	1.26
Helium	B	Normal element displaced	12.04	21.198	53.843	21.507	54.628	-40.2	.054	1.33
Capsule	[a]B	Normal	.54	21.143	53.703	21.176	53.787	-4.3	.115	1.13

[a]One type B fuel plate (1 by 0.875 by 0.1 in. or 2.54 by 2.27 by 2.54 cm).

were to occur, the reactivity would change from -38.7 to -18.2 cents; a change of 20.5 cents.

Error Analysis

Observation of approximately 1000 data points showed that the data for the element power distribution were within ±4 percent of the curves presented 90 percent of the time. Table II gives a sampling of the error data according to the experiment performed. The thermal-neutron flux data on the average were much better on the basis of an error analysis. The data were within ±4 percent of the curves presented 100 percent of the time. The values of reactivity had an uncertainty of ±1.5 cents associated with them.

TABLE II. - SAMPLING OF ERROR DATA

Inpile tube mockup	Element type	Confidence level, percent	
		For ±2 percent error	For ±4 percent error
Helium	A	57	86
Helium	B	74	97
[a]Helium	B	63	89
Argon	A	57	91

[a]Enclosed in tungsten-molybdenum box.

Survey of Results

A tabular summary of the results considered to be pertinent by the experimenters with either the helium, argon, or capsule inpile tubes are given in table I. A few of the more general results of the experiments are the following: For the helium inpile tube data of table I, as the amount of uranium 235 per unit length is increased, the average power decreases and the peak-to-average power increases, as does the reactivity of the helium inpile tube. This same trend is evident for the argon inpile tube and for the capsule inpile tube.

SUMMARY OF RESULTS

In an investigation to obtain nuclear design data for experiments to be conducted in the Horizontal Through Hole 1 (HT-1) of the Plum Brook Reactor, experiments were conducted in HT-1 of the Plum Brook Mockup Reactor (MUR). In the MUR experiments with the helium, argon, and capsule inpile mockup assemblies, the following results were obtained:

1. The thermal-neutron fluxes measured in HT-1 and along the inpile tube axis at the core north-south vertical midplane with the helium, argon, or capsule inpile tube in HT-1 were 3.02×10^9, 1.83×10^9, and 3.37×10^9 neutrons per square centimeter per second per kilowatt of reactor power, respectively. These results were obtained for the core in a mixed configuration. The core configuration did affect the magnitude of the thermal-neutron flux.

2. The average fission power production per gram of fuel in either of the inpile tube facilities decreased with the addition of fuel per unit length. Thus, the average fission power production per gram of fuel was lower for the type A element than for the type B

element. The average fission power production for the elements tested ranged from 0.04 to 0.10 watt per gram of uranium per kilowatt of reactor power.

3. The distribution of gamma heating was similar to that of the thermal-neutron flux distribution in the helium inpile tube. The gamma heat generation ranged from approximately 0.5×10^{-4} to 1.0×10^{-4} watt per gram per kilowatt of reactor power.

4. The reactivities of the helium, argon, and capsule inpile tubes were -40.2, -20.2, and -4.4 cents, respectively. Addition of uranium to HT-1 gave a positive change in reactivity. The two accident conditions studied (element holder failing and inpile tube flooding) caused changes in reactivity of -1.4 and 20.5 cents, respectively.

5. The peak-to-average power increased as the number of grams of uranium 235 per unit length was increased.

Lewis Research Center,
National Aeronautics and Space Administration,
Cleveland, Ohio, August 15, 1967,
120-27-04-55-22.

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